NON-PUBLIC?: N
ACCESSION #: 8905300041
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 of 6

DOCKET NUMBER: 05000410

TITLE: Nine Mile Point Unit 2 Reactor Scram Due to Turbine Trip
Caused by Loose Wire Connections
EVENT DATE: 04/13/89 LER #: 89-014-00 REPORT DATE: 05/15/89

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION
50.73(a)(2)(iv)
OTHER: Special Report

LICENSEE CONTACT FOR THIS LER:
NAME: Robert G. Smith, Operations
Superintendent, Unit 2 TELEPHONE: (315)349-2388

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: FG COMPONENT: JX MANUFACTURER: E355
X EA BKR G080
REPORTABLE TO NPRDS: N
N

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On April 13, 1989 at 1101 hours with the reactor mode switch in RUN, and the reactor at 100% rated thermal power, Nine Mile Point Unit 2 experienced a reactor scram. This event was a result of a turbine trip due to the actuation of the generator protection circuitry. The generator protection circuitry initiated a fast transfer of house service loads to the station reserve transformers. One of the 13.8kv electrical buses failed to transfer which caused a loss of feedwater. Reactor water level decreased as the turbine control bypass valves modulated to control reactor pressure, which caused the automatic actuation of the High Pressure Core Spray (CSH) and Reactor Core Isolation Cooling (ICS) systems.

The cause for the turbine trip was determined to be a disconnected wire located in the main generator potential transformer cubical. This created a

signal to the tripping and alarm relays causing a turbine trip. The cause for the 13.8kv bus fast transfer failure was determined to be the positive interlocking roller for the breaker not fully engaged.

Corrective actions include: (1) Relanded the disconnected wire and tightened other potential transformer connections; (2) Revising the electrical maintenance procedure (N2-EPM-GMS-R693) to check connection integrity in high vibration areas; (3) Issuing a lessons learned transmittal outlining the problems encountered during this event.

END OF ABSTRACT

I. DESCRIPTION OF THE EVENT

On April 13, 1989 at approximately 1101 hours, Nine Mile Point Unit 2 (NMP2) experienced a reactor scram due to a turbine trip. At the time of the event the reactor mode switch was in "RUN" (Operational Condition 1) with the reactor at 100% rated thermal power, and the reactor pressure and temperature at 1003.5 pounds per square inch gauge and 546 degrees Fahrenheit respectively.

The turbine trip resulted from a generator protective circuit relay actuation. The turbine trip initiated a fast transfer of house service loads from the station normal service transformer to the station reserve transformers. Switchgear 2NPS-SWGOO3 failed to transfer. This caused a loss of feedwater since the operating feedwater pumps (FWS-P1B and FWS-P1C) were being powered from 2NPS-SWGOO3. The complete loss of feedwater coupled with the normal operation of the turbine bypass valves (TBV's) to control reactor pressure caused reactor water level to decrease to the Level 2 (108.8 inches) setpoint. The lowest the reactor water level reached during the transient was approximately 98 inches. The level 2 setpoint caused the automatic initiation of the High Pressure Core Spray (CSH) and Reactor Core Isolation Cooling (ICS) systems, which injected water from the Condensate Storage Tanks (CST) to the reactor vessel to restore water level. NMP2 entered into an Unusual Event and its Emergency Plan based on Emergency Core Cooling System (ECCS) injection on a valid initiation signal.

When reactor water level was increased to normal, CSH injection was secured. ICS injection was automatically secured at Level 8 (202.3"). Operators continued to monitor reactor water level and believed that all vessel injection was secured.

However, feedwater was continuing to be injected. Power was lost to motor operated feedwater regulation valve 2FWS-LV10B, causing the valve to fail as is. This condition was not recognized by the operator since the indication of the valve position and the position demand failed downscale. The operator believed 2FWS-LV10B had closed as had 2FWS-LV10C, the complementary hydraulically operated regulation valve. Therefore, with the combination of cold water injection, steam line drains, and steam consumption by ICS, reactor pressure was lowered to the point where condensate booster pump discharge pressure exceeded reactor pressure. These conditions permitted feedwater flow to the vessel through the open LV10B valve. The operator recognized that feedwater flow was increasing causing reactor water level to increase and informed the Station Shift Supervisor of these conditions. The Station Shift Supervisor then ordered the remaining condensate booster pump secured. The maximum reactor vessel water level recorded during the transient was approximately 258 inches. Water level then decreased due to boil off. When Level 8 had cleared its setpoint the ICS pump was used to maintain water level. The CSH pump was tripped after it was verified to be no longer needed to maintain reactor vessel level.

The failure of the 13.8kv switchgear (2NPS-SWGOO3) to transfer was caused by a "Positive interlocking roller" not being fully engaged. This positions a limit switch, which provides a breaker position interlock to the closing circuit.

I. DESCRIPTION OF THE EVENT (Cont'd)

Complications occurred during the event when a licensed operator inadvertently de-energized the remaining 13.8kv power board (N2-NPS-SWGOO1). The licensed operator observed a "Red Flagged" condition with no position indicating light (due to a loose light bulb). The operator attempted to check voltage on the affected board but observed the wrong meter (which indicated zero). When the control switch for breaker 1-1 was placed in the reset (or lockout) position, power board N2-NPS-SWGOO1 was de-energized. The operator immediately realized his mistake and re-energized N2-NPS-SWGOO1 in accordance with station procedures.

During the momentary loss of 13.8kv power to the house service loads, the remaining Circulating Water (CWS) pumps were de-energized. The decrease in the Main Condenser water box level after the loss of power to the CWS pumps, prevented the immediate restart of the CWS pumps and the subsequent loss of the Main Condenser as the primary heat sink. The SSS directed the use of the Steam Conden
ing mode of the Residual Heat Removal (RHS) system loop A. The Main Steam Isolation Valves (MSIV's) were directed to be closed as the Main Condenser vacuum decreased to 10 inches of mercury. As the inboard MSIV's were being closed, a Group 1 Isolation was received due to low condenser

vacuum and a fast closure signal for the MSIV's was received (closing all the MSIV's). The ICS system was used to maintain reactor vessel water level and the Steam Condensing mode of RHS was used to maintain reactor vessel pressure control. A reactor cooldown to ambient conditions was then performed. Uninterruptible Power Supply 1D (UPS-1D), tripped due to an overload condition. This resulted in a loss of approximately one half of the Gai-tronics system in the plant, a total loss of Gai-tronics in the Control Room (affecting communications with plant operators outside the Control Room) and a partial loss of emergency lighting.

II. CAUSE OF THE EVENT

A root cause analysis was performed using Site Supervisory Procedure S-SUP-1, "Root Cause Evaluation Program". The root cause for this event was determined to be loose wire connections in the Main Generator Potential Transformer cubical, 2GMS-CUBO1, for circuit 2SPGZO3. This is attributed to poor installation compounded by vibration in the area of the connections.

III. ANALYSIS OF THE EVENT

This event is reportable wider 10CFR50.73(a)(2)(iv):

"Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including RPS, that resulted from and was part of the preplanned sequence during testing or reactor operation need not be reported."

The reactor scram was due to a turbine trip, which was in response to the actuation of a generator protection relay. The reactor scram is a protective function and therefore poses no adverse safety consequences. The failure of the 13.8kv power board (N2-NPS-SWGOO3) to transfer and subsequent loss of the remaining 13.8kv power board (N2-NPS-SWGOO1) did not pose a threat to the health and safety of the General Public as the three (3) divisions of Emergency Core Cooling Systems (ECCS) were operable with power sources from off-site and Diesel Generators. Only two of the three ECCS divisions are utilized to achieve safe shutdown.

CSH and ICS automatically initiated at Level 2 (108.8 inches), to restore level as designed.

The problems that were encountered with the overload condition of UPS-1D (partial loss of communications outside the Control Room and partial loss of emergency lighting) did not compromise the safety of the general public as the safe shutdown of the plant can be achieved from the Control Room.

An evaluation of 10CFR50 Appendix R, Section III.J, "Emergency Lighting" requirements was conducted. The evaluation concluded that there is no in|pact on the Appendix R safe shutdown analysis.

Engineering is, however, conducting a review of associated essential light circuits to reconfirm that cable routings do not traverse openly in postulated fire areas (i.e., UPS-1D circuit cables through the Control Room and Relay Room areas). If unanalyzed conditions are found, a supplement to this report will be issued.

Transient recording indicated that water level was slightly above the lowest elevation of the Main Steam Line (MSL) nozzle. However, the level trends indicate that water level did not reach the Main Steam Line (MSL). A firm conclusion that water did not flow down the Main Steam Lines could not be made. However, if water flowed down the Main Steam Lines, it was for a very short duration. Based on prior Engineering analyses performed on previous vessel overfill, the effects of potential transients created, if water had entered Main Steam Lines, were within plant design margins.

III. ANALYSIS OF THE EVENT (Cont'd)

In accordance with the requirements of Technical Specification Sections 3.5.1(f) and 6.9.2, Emergency Core Cooling System (ECCS) Injections, the following data is provided:

For the HPCS nozzle,

- Total accumulated initiation cycles to date = 4
- Current usage factor value remains below 0.70

The duration of the event was approximately 8.5 hours from the time the event was initiated (1101 hours) until the Shutdown Cooling Mode of RHS was controlling reactor temperature and pressure (approximately 1930 hours).

IV. CORRECTIVE ACTIONS

1. Relanded the disconnected wire and tightened the other potential transformer connections.

2. The Electrical Preventative Maintenance Procedure N2-EPM-GMS-R693 will be revised to ensure the integrity of the wire connections in the areas of high vibration is checked.

3. The Electrical Preventative Maintenance Procedures will be revised to check interlock mechanism for proper operation.

4. A Lessons Learned Transmittal was generated by the Operations Department to discuss the problems associated with this event.

5. The licensed operator involved in de-energizing N2-NPS-SWGOO1 was counseled on self-verification and communication.

6. Operation Procedures have been revised to ensure visual verification of the interlock roller positions on 13.8kv breakers.

7. Operation Procedures have been revised to ensure that Feedwater, Condensate Booster and Condensate pump power supply lineups are separate for running pumps.

8. Operation Procedure was revised to provide improved direction for water level control following a reactor scram.

9. The electrical loads on UPS-1D have been reduced to prevent a trip on overload. Additional modifications are being considered to further reduce UPS-1D loads.

V. ADDITIONAL INFORMATION

A. Failed Components:

1. Failed Component Identification: 2VBB-UPS1D
Component Description: Uninterruptible Power Supply for
Station Lighting/Communication
Component Vendor: Exide Power Systems Division

2. Failed Component Identification: 2NPS-SWP-003-1
Component Description: 3000 Amp Breaker
Component Vendor: General Electric Company

B. Nine Mile Point Unit 2 has not experienced a reactor scram caused by a similar event.

NM NIAGARA NMP49057
MOHAWK

NINE MILE POINT NUCLEAR STATION/P.O. BOX 32 LYCOMING, NEW YORK 13093/TELEPHONE
(315) 343-2110

May 15, 1989

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 89-14

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report:

LER 89-14 Is being submitted in accordance with 10CFR50.73(a)(2)(iv), "Any event or condition that results in manual or automatic actuation of an Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

A 10CFR50.72(b)(2)(ii) report was made at 1150 hours on April 13, 1989. This report was completed in the format designed in NUREG-1022, Supplement 2, dated September 1988.

Very truly yours,

L. Burkhardt III
Executive Vice President
Nuclear Operations

LB/GB/mjv
(0492V)

Attachment

cc: Regional Administrator, Region 1
Sr. Resident Inspector, W.A. Cook

*** END OF DOCUMENT ***